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Abstract

TRENDS IN WATER-MODERATED WATER-COOLED POWER REACTOR DESIGN

A.Ya.Kramerov, V.V.Stekolnokov
et al

A designer of water-moderated water-cooled power reactors should be guided by the considerations of maximum economic efficiency of a reactor. The efficiency of a reactor largely depends on reactor power per unit of reactor volume, simplicity and reliability of reactor unit design and the degree of optimalization of the operating parameters.

The paper discusses some considerations for selecting ratings and design parameters for water-moderated water-cooled reactors of progressive design and presents certain analytical techniques for parameter optimalization.

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Among other aspects the authors discuss problems involved in design of the steam generation system and selection of steam parameters in the power-generating cycle, types of steam generators and reactor body and present their views on the structure of the reactor core in connection with the necessity of frequent refueling of the reactor and the location of the reactor channels of the protection control system.

Particular attention is paid to simple approximate analytical procedures for the optimalization of the basic parameters of large water-moderated water-cooled reactors, viz. selection of a housing diameter, the geometry of the core cell, heat fluxes and specific heat transfer, dimensions and power rating of the reactor core, parameters and flow rate of a heat-transfer agent, thermal stress, steam pressure in the steam generator, etc.

As a promising system for reactor design the paper describes a water-moderated water-cooled reactor with individual temperature controllers at fuel units, deep burnup and refueling under pressure.

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Abstract

STARTING AND ADJUSTMENT OF WATER-MODERATED WATER-COOLED POWER REACTOR AT NOVOVORONEZHSKAYA POWER STATION

S.A. Skvortsov, G.L. Lunin, Ovchinnikov et al.

The paper delineates the basic procedures followed in the starting and adjustment programme for water-moderated water-cooled reactors.

The basic stages of adjustment and starting of the reactor and the auxiliary equipment are described and some improvements and modifications carried out during its adjustment are discussed.

The analytical values of physical, thermal and hydraulic parameters are compared with the variables measured during testing and starting of the reactor and during the period of reaching full rated capacity.

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Abstract

EXPERIMENTAL PRESSURE VESSEL TYPE REACTOR FOR STUDYING PROBLEMS OF BOILING AND VAPOUR SUPERHEATING

Vorobjev, I.N.Sokolov,
A.A.Khokhlatchev, V.N.Gritzkov
et al

In the USSR, construction work is being completed on an experimental reactor of the pressure vessel type to study problems of boiling and nuclear superheating of vapour inside the reactor.

The installation scheme of a universal type is considered in which it is possible to conduct investigations on boiling and vapour superheating problems, within a wide range of heat transfer medium parameters.

The reactor provides an electrical capacity of approximately 50 megawatts; it is possible to obtain saturated vapour with pressures ranging from 30 to 100 ata, and also vapour with a pressure of 90 ata and nuclear superheating of up to 500°C.

The layout of the vessel and the active zone of the reactor designated for operation with fuels of various enrichment and for various parameters of the heat-transfer medium are described. The research and engineering work program to be carried out on the experimental installation is discussed in detail.

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Abstract

ORGANIC MODERATED NUCLEAR POWER PLANT

Yu.Aleksnko, K.K.Polushkin, N.V.Zvonov,
V.D.Tetyukov et al

The 750 kW(e) nuclear power plant for use in remote, difficult--to-reach areas is considered.

The discription of the plant is given in the report. It cosists of sepa rate modules (each weighing up to 20 tons), their mounting takes two or three months. An organic fluid (hydrostabilized gas oil) is used as a coolant and moderator.

The reactor and fuel elements are described.

The reactor and the primary circuit equipment are made of carbon steel. There is no biological shielding at the plant(except the reactor).

A system for coolant regeneration is described.

The reactor start-up and the nuclear power plant operation at loads are described.

The nuclear power plant was brought to 100% power on August 11, 1963.

Some information on plant test operation is presented. The results are in a perfect agreement with design criteria.

It is shown that the successful operation of the plant permits construction of more powerful stations with an organic cooled and moderated reactor.

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Abstract

THE BELOYARSK NUCLEAR POWER STATION FIRST UNIT PILOT OPERATION

A.N.Grigoryants, P.I.Aleschenkov,
L.A.Kochetkov, Nevskiy et al.

Pilot operation of the first unit of the Beloyarsk nuclear power station is described. The first unit comprises a uranium-graphite tube type power reactor, steam generator which produces saturated steam and high pressure turbogenerator operating on nuclear superheat steam.

The main feature of the first unit start-up is that the start-up and in particular reactor heating from a cold state is obrained without a heat external source using only reactor inherent heat of uranium-235 nuclear fission.

This feature makes it somewhat difficult to put the reactor into operation but at the same time this is absolutely necessary for a superheat nuclear power station.

Research and experimental work before and during reactor pilot operation is discussed, as well as simultaneous checking-up of a power plant equipment, operation at loads, the steady-state unit early stage operation and setting-up of separate systems, control and measuring devices and automation instruments are pointed out.

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Abstract

DEVELOPMENT OF SUPERHEATING POWER REACTORS OF BELOYARSK NUCLEAR POWER STATION (БАЭС) TYPE

N.A.Dollezhal, P.I.Aleschenkov, I.Ya. Emelyanov, A.D.Zhirnov, G.A.Zvereva, N.G.Morgunov, K.A.Kryukov, Yu.I.Mityayev, G.D.Knyaseva
K.A.Kryukov

Development of heat removal systems and technological flow sheets of a power station with direct cycle uranium-graphite tube type reactors and nuclear superheating of steam is discussed.

Succession of the development of this most promising type of modern direct cycle supercritical up to 800 Mw(e) reactors, direct descendants of the First Atomic Power Station of Academy of Sciences of the USSR, is shown.

The problems encountered during design and experimentation with the 200 Mw plant (2nd БАЭС unit) are considered.

Fuel element and channel designes, their position in the graphite moderated core and operation conditions, lattice spacing as well as meny features of the reactor concepts characteristic of different coolant flows and power ratings are discussed.

Increase in electrical, hence thermal power of the plant cause each fuel channel power rise.

The single fuel channel power rise is limited by:

- critical heat fluxes for steam generating tubes;
- maximum permissible fuel element can temperature in

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steam generating tubes.

This necessitates the development of more promising fuel channels and auxilliary equipment.

Main features of direct cycle reactor concept, namely feed water regime and steam reactivity problems, are mentioned.

Flow diagrams are described, and direct cycle advantages over those of indirect cycle concept are pointed out.

In conclusion, economics of the considered concepts is evaluated.

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Abstract

PORTABLE ATOMIC POWER STATION

I.F.Bychkov, G.N.Ushakov, Yu.A.Sergeev
et al

The description is given of portable power station TES-3 (ТЭС-3) being now in operation in Obninsk. TES-3 main equipment fits on four wheeled truck-trailers. The station is designed according to two-circuit diagram with water-moderated heterogeneous II Mwt reactor cooled with water at 130 atm. In once-through steam generators there is produced an overheated steam at 20 atm and 285°C, its amount being enough for the power output ~ 2000 kWt on the turbine shaft. The amount of water 900 t/hour is necessary to cool the condenser. The core lifetime lasts about a year.

The equipment, reactor including, can be transported by means of wheeled truck- trailers to a new location of operation. For this purpose the reactor is provided with the shielding revealing the possibility to dismantle and transport it together with partial or total burn-up core in several hours after shut-down. When moved, the reactor is cooled down by means of the air radiator.

To provide the shielding from the irradiation, when the reactor is in operation, there is an additional shielding consisting of local materials (reinforced concrete, sand). The reactor is overloaded with the help of 25-tonn autocrane under field conditions, the reactor vessel top is not removed.

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Abstract

SODIUM COOLED FAST REACTORS

A.I.Leipunskii, Koshkin et al

On the base of investigation and operational experience of experimental fast reactor BR-5, now in the Soviet Union the prototype of full scale nuclear power station with fast reactor 350 Mwe is being worked out.

This paper describes the reactor with core Φ I500mm and H=I000mm, and power I000Mwt. Fuel elements are the stainless steel tubes Φ 6,Imm, S=0,4mm, filled with dioxide uranium and plutonium wuth average density 8g/cm^3. Flattening of the power distribution in the reactor was carried out by means of various enrichment of the fuel I3% (in the internal region) and I9% (in the external region).

The fuel elements are ribbed for spacing and inserted into subassemblies.
The radial blanket 600mm thick consists of cylindrical elements filled with depleted uranium oxide is inserted into subassemblies of the same type.

The reactor is controlled by elements with B_4C, reactivity change due to burnup is compensated by movable subassemblies with fuel elements.
Reactivity changes due to heat up and power rise up to rated value are compensated by subassemblies with B_4C elements. Similar B_4C subassemblies are used for the scram of the reactor.

In the report there are main physical data and discription of the principle systems and equipment (shielding, fuel charge and discharge mechanism, pumps, heat exchangers, steam genera-

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tors) and coolants parameters of the primary, secondary and third loops (systems).

This reactor is the first full scale power reactor of the type plant for commissioning by 1970. Operational experience of the reactor and main equipment will permit to start construction more power full and modern reactor, capable of long continuous operation as a source of electrical power. Preliminary studies of such reactor 1000Mwe showed, that the further increasing D/H is desirable.

The reactor with monocarbide as a fuel has volume 3000e. It's internal breading ratio is equal to unity and enables necessary long operatuin.

Main dates of the reactor are given in the report.
As a fuel of nuclear reactors sintered mixture of dioxide uranium and plutonium is considered on the first place. In the first reactors dioxide of enriched uranium will be used.
The possibility of the application uranium and plutonium carbide as a reactor fuel, as well as metalic fuel is studied.

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Abstract

5-YEAR OPERATION EXPERIENCE ON THE FAST REACTOR

O.D.Kasachkovskii, N.M.Aristarkhov et al.

Principal results of 5-year operation of sodium-cooled fast reactor BR-5 (nominal power 5000 kw) are given in this report.

Reactor operation affter achieving the design burnup (2%) sith some fuel elements being untight is described. The results of contamination analysis of coolant, technological equipment and pining surfaces, and reactor gas blanket by fission products are given. Information on the main technological equipment operation is reported. Untightness that took place in some fuel elements was shown not to result in any appreciable difficulties during the operation of predesigned systems.

Special devices for investigation (during reactor operation) of fission products released into coolant and reactor gas blanket, and their deposition over pipe inner surfaces are deseribed. Spectrometric analysis indicated the residual radioactivity of primary coolant to be caused by Cs^{137}. It is pointed out that coolant contamination may be decreased by means of cold traps.

The experiments conducted proved that only inert gases (Xe and Kr isotopes) were released during reactor operation into gas blanket. Iodine was not detected.

The results of radiochemical examination of coolant and

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deposits (on pipe inner surfaces) sampling are given. The results of investigation of plutonium oxide fuel elements with burnup more 5,5%.are described, the causes of fuel element canning rupture are analyzed.

Some results of the kinetic study by means of step-function method are described. The results are compared with the oscillation method data obtained. Transient characteristics of power to reactivity for different coolant flow rates were obtained.

Some uranium monocarbide fuel elements were loaded into the reactor BR-5 core to pass to a new fuel material by the end of I964.

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Abstract

OPERATION OF ATOMIC PLANT ON THE ICEBREAKER LENIN

A.P.Aleksandrov, N.S.Khlopkin, B.C.Pologikh, N.M.Tsarev, A.K.Sledzjuk et al.

The paper summarizes the experience gained in the course of operation of the atomic plant on the icebreaker LENIN during 1959 to 1965 in the Arctic region under various conditions: when taking caravans of ships through ice (the total number of the ships is about 300), when sailing at high latitudes to bring the personnel and equipment of the drifting station "North Pole 10" to their base, when sailing in clear waters at roll angles as great as 30°. Under all these conditions the atomic plant provided an uninterrupted supply of energy for the icebreaker with the reactors operating at various power levels including the maximum power of 90 MW.

The paper presents data on the changes in the neutron-physical and thermotechnical characteristics of the reactor cores throughout the company which somewnat exceeded the rated duration. Such data cannot be obtained from the experiments with the fuel elements in the loops of the research reactors. They become available only as a result of the complete burn-up of the core with the specific positions of the control components and at specific distortions of the neutron fields.

The operational experience has shown that the design of the atomic plant and its arrangement on the icebreaker proved to be successful and equipment reserves quite sufficient. Recommendations are made for the modernization of some components and systems on the basis of the four-year experience. It has been shown that the required water conditions in the primary and secondary circuits can be maintained with the aid of simple mechanical and ion-exchange filters

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provided appropriate steps are taken to minimize the possibility of sea water leakage into the equipment and in the pipeline joints. No overirradiation of the personnel servicing the atomic plant took place during operation. The plant proved to be so reliable that a daily inspection of the periodically serviced rooms was quite sufficient.

The radiation safety was enhanced by the concentration of the atomic plant in a special isolated compartment fitted with an independent ventilation system, the equipment for the removal and temporaty storage of the active water, temporary solid waste storages. Radiation safety measures are described for certain disturbances in the normal operation of the plant.

The paper also describes the maintenance of the icebreaker at the base.

As a whole, the operational experience has shown that the use of atomic plants on icebreakers has highly promising prospects.

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Abstract

SOME OPERATIONAL RESULTS OF FIRST POWER STATION AS EXPERIMENTAL SET - UP

Ushakov G.N. et al

Designing of new nuclear power plants and in the first place Belojarskaja atomic power station raised a number of problems required the development of special experiments such as: carrying out water-boiling and steam superheating operating conditions in tube fuel elements, studying conditions of formation, accumulation and release of radiolytic O_2 and H_2 and graphite and steel behaviour in pile.

I. The primary problem of new plant construction is the experimental development of fuel elements. This problem required the establishment of various experimental loops at the first atomic power station:

1) two-contour steam-superheat loop,
2) natural-circulation water loop,
3) water-unit for chemical experiments,
4) high-pressure water unit,
5) high-melting organic coolant loop,
6) low-melting organic coolant loop.

2. High-temperature coolant and large power density in fuel elements resulted in high temperature of graphite core. Commercial N_2 is used for graphite core filling. Contents of O_2, H_2 and CO_2 is maintained by I%.

The feasibility of long-lived graphite core operation up to 700 - 800°C is confirmed by ten years successful work. The measurements of physical and mechanical properties of irradiated technological channel graphite blocks were made. Graphite core operational conditions were observed to be sa-

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tisfactorily after ten years work. Irradiated and nonirradiated graphite behaviour in N_2, and N_2, H_2O environment was studied in the range of 700, 900, I000 and II50°C temperatures.

3. The channel and fuel element design was subjected to long time exposure to neutron radiation and high heat flux being rather stable during steady-state and transient operation of atomic power station. The tube fuel element channel design is versatile and permits channel to operate under boiling and steam-superheating. No getting fissian products is possible into the primary contour in the above mentioned design even under the failure. Experiments resulted in the definition of allowable burn-up U^{235} were conducted. In some channels the lifetime fuel exposure up to 30-32 kg/1tonn was reached within 44000 hours.

4. New raised problems of experimental channel testing put a question about some reconstruction of reactor and its system. Circular system reconstruction of the first contour was accomplished in I963. Pumps with hydrolic shaft seals were replaced by cenned rotor ones. To extend experimental possibilities the diameter increasing in some charge holes of graphite core up to 95 - I20 mm was supposed.

5. The results of investigation of water and superheated steam radiolysis and also superheated steam radioactive impurities deposition on turbine vane models are discussed. This work was carried out on the loop simulating Belojarskaja atomic power station schemes. It was determined that radiolitical gas yield strongly depends on O_2 dissolved in water. Deposition radioactivity on turbine vane models was I0% initial in a month after shut-down. In depositions were found out Cr-5I, Fe-59, Ca-60, Ni-58, Ca-45.

Radioactive depositions are chemical stable compounds wnich are not affected by water and steam.

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Abstract

HIGH-TEMPERATURE GAS COOLED POWER REACTOR

N.N.Ponomarev-Stepnoy, B.N.Kolbasov,
A.N.Viewnov et al.

The paper discusses the prospects for utilization or gas-cooled reactors in atomic power industry of the USSR. Reactors with different coolants are analysed. The possible use of single-circuit gas-turbine plants at high coolant temperatures is discussed along with a double-circuit installation with a steam-turbine cycle.

The paper describes a gas-cooled power **reactor,** and eleborates calculation techniques and methods of selection of optimum neutron physical and thermal characteristics of the reactor and parameters of the thermodynamic cycle.

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Abstract

ATOMIC POWER PLANTS IN THE GENERAL POWER SYSTEM OF THE USSR

K.D.Lavrinenko

The present power development in the USSR is characterized by quick increasing of power capacities through commitment of large thermal and hydro power stations which have great power units as well as through integration of separate power systems into enormous power complexes.

The availability of practically unlimited resources of coal, oil, gas and hydropower in the USSR makes it possible during a certain period of time to build only pilot-commercial atomic power plants with various reactors in order to gain some experience in designing, building and operation of such power plants.

The enormous scale of power development provided in the program of the Communist Party of the Soviet Union and the unevenness in the distribution of electric power consumption districts and the irregularities of power source deposits will demand in the future the building of a number of large atomic power plants mainly in the districts where the resources of fuel and hydropower are limited.

The paper discusses some permises of integration of atomic plants into power systems of the USSR and the main requirements the atomic power plants are to meet in order their wide integration into these systems was economically warrant.

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Abstract

THEORETICAL AND EXPERIMENTAL INVESTIGATIONS OF THERMIONIC METHOD OF DIRECT HEAT - TO - ELECTRICITY CONVERSION

I.I.Bondarenko , I.P.Stakhanov et al

I. Possibilities of application of thermionic heat-to-power converter in nuclear engineering are discussed for middle- and low-power nuclear reactors.

2. The results of theoretical and experimental investi - gations of fundamental converter processes in various operating conditions (low-pressure with charge neutralization, low-mean-free-path limit, low-voltage-are limit) are presented. Physical and technical features of these operating conditions are considered.

The results of investigations of surface ionization and thermoelectron emission properties for main refractory and design materials are also presented, including characteristics of therminonic low-pressure converter with charge neutralization. Thermoemission constant values are briefly discussed.

In the report there are considered theoretical and experimental results concerning voltage-current characteristics of the converter in the low-mean-free-path limit, saturation-current and optimal power dependence on the electrode temperature, gas pressure and emission properties of electrodes. The heat-flux carried over from a cathode by electrons and the converter efficiency are calculated. Some physical parameters of the con-

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verter are obtained on the basis of comparison of theoretical and experimental data.

The investigation of the converter in low-voltage-arc conditions is of special interest. Voltage-current characteristics, power output and their dependence on emission parameters, electrode temperatures and gas pressure are presented. Breakdown voltage dependence on gas pressure and cathode temperature is given. Results of probe measurements of electron temperature, density and potential distribution for low-voltage-arc in cesium vapours are also presented.

The influence of external magnetic field on the thermionic converter and low-voltage-arc is investigated. The dependence of voltage-current characteristics and breakdown potential along and across the magnetic field are presented. In addition there was investigated also (theoretically and experimentally) the dependence of saturation-current and voltage-current characteristics of the converter on magnetic-field-strength.

There are given results concerning theoretical and experimental examinations of current fluctuations of the converter. Conditions resulted in appearing fluctuations and dependence of fluctuation frequency on different physical parameters are considered. The reason of appearing these fluctuations is studied theoretically.

3. There are considered fundamental scientific and technical problems needed to be solved to create thermionic emission converter with a nuclear reactor as a source of heat.

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Abstract

THERMOELECTRICAL GENERATOR WITH ISOTOPIC HEAT SOURCE

I.G.Gverdtsiteli, Z.B.Ershova, Yu.D.Gubanov et al

The design of thermoelectric generator operating in vacuum is presented in the paper. The description of the isotopic heat source of the thermoelectric generator on the high temperature thermoelectric material base.

The test method of the device and the test stand desciption are given.

The stand test results (volt-amper characteristics, temperature distribution, the device activity and so on) are presented in tables and diagrams.

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Abstract

NUCLEAR REACTORS FOR STEAM GENERATION AND DISTRICT HEATING

Yu.I.Koryakin, A.A.Loginov

Combined electricity production and heat generation based on dead steam use may significantly improve performance and economics of nuclear power stations because about 5/7 of live steam enthalpy at mean pressure is lost with the condenser cooling water. The advantage of a nuclear central district heating plant over an ordinary one is that it may be used for heat supply of remotedly sited industrial enterprises which main economic characteristics (power and heat generating costs) little depend upon the place they are situated at.

The possibility of heat supply over great distances as well as ever increasing reliability and safety of nuclear power plants allow us to admit that nuclear district heating plants for industrial and house heating may be used in the nearest future.

At present in the USSR there is some operation experience of nuclear reactors for combined electricity production and district heating as well as for district heating only (heating loop at the Beloyarsk nuclear power station, organo-organic experimental facility of the Atomic Energy Institute after I.V.Kurchatov etc.).

The possibility to use nuclear reactor for steam generation and district heating is analysed and it is shown that nuclear reactor plants can and must be used for district heating purposes.

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Abstract

REACTOR SM-2 WITH THE HIGHEST AVAILABLE NEUTRON FLUX

S.M.Feinberg, V.A.Tsykanov,
E.D.Vorobjev et al

The paper describes the main experiments conducted to ascertain separate parameters of the SM-2 reactor and also some problems of its operation. The experimental results obtained are compared with the theoretical calculations. Some design changes made in the period of adjustment and operation of the reactor are also described.

The following questions are discussed in the paper: 1) dependence between the critical charge of the reactor and the confiquration of the core; 2) change in reactivity as the uranium-235 burns out; 3) change in reactivity after the replacement of burnt-up fuel units by fresh ones; 4) control rod effectiveness; 5) neutron spectrum; 6) slow-neutron distribution throughout the reactor volume; 7) effects of various materials placed in the central cavity and other vertical channels on reactivity and slow-neutron distribution; 8) poisoning, etc.

Experiments were conducted both in the reactor and on a special physical stand. Critical experiments were mainly carried out on the stand. The neutron spectrum was studied with the aid of special small-size pulse chambers containing uranium-235, as well as uranium, indium and other tracers. The unevenness in power release depending on the core height was measured according to the residual gamma-activity of the fuel ele-

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ments.

Physical calculations were made in various approximations. The spectrum of slowing-down neutrons and the volume distribution of various-energy neutrons were calculated in P_1 -approximation. Calculations connected with the determination of critical dimensions, change in reactivity as the uranium-235 burnt out and after the reloading of fuel assemblies, volume slow neutron distribution, etc., were made in two-group two-dimensional approximation. The slow-neutron spectrum was calculated for a homogeneous medium. The results of the calculations on the whole satisfactorily agree with the experiments.

The authors describe their experience in operating some engineering assemblies of the reactor: system of protection control, the reloading machine, etc. Some data on the stability of some construction materials are given. Experience in operating the reactor made it possible to ascertain which of the units in the reactor design are more or less good. In particular, the fuel elements proved to be quite good.

The reactor showed the following characteristics:

1. Power capacity - 50,000 kW.

2. Thermal neutron flux in the "neutron trap" - $(2.5 \pm 0.5).10^{15}$ n/cm.^{2}sec.

3. Fast neutron flux with an energy of more than 1 Mev. in the core-about 10^{15} n/cm^2.sec.

4. Maximal thermal loads on the surface of fuel elements-6.10^6 k cal/m^2.hour.

5. Maximal thermal load per unit volume of the core-$4.5.10^3$ kW/l.

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Abstract

CONSTRUCTION OF POWERFUL LOOP-TYPE REACTORS
THE МИР RESEARCH LOOP-TYPE REACTOR

Yu.M.Bulkin, A.P.Bovin,
Yu.G.Nikolaev et al.

The paper consists of two sections.

The first section compares modern loop-type reactors and deals with the main engineering and physical problems of their consturction and operation.

Four types of powerful loop-type reactors are compared:

a) vessel-type reactors;

b) vessel-type swimming-pool reactors;

c) channel-type reactors;

d) channel-type swimming-pool reactors.

In the second section, the Мир research loop-type reactor, now being constructed in the Soviet Union, is described; it will be a channel-type swimming-pool reactor.

The following problems characteristic of loop-type reactors are dealt with in the paper:

a) correspondence of conditions for testing fuel elements in a loop-type reactor to the natural working conditions of the-

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se elements;

b) maintenance of the prescribed conditions for testing fuel elements simultaneously in several loop channels;

c) increase in the duration of loop-type reactor working cycles ;

d) rapid re-starting of a loop-type reactor after a spurious trip;

e) reduction of the thermal neutron flux depressions where the loop channels are installed;

f) choosing types of loop channels and the place for them in the reactor;

g) safety of channel tests;

h) preparatory work preceding fuel element tests in a loop-type reactor.

The maximum power of the Мир -type reactor, referred to in the paper, is 100 megawatts. To maintain the prescribed conditions for testing fuel elements in the loop channels of the reactor its power can in the course of operation be changed. Highly enriched uranium is to be used as fuel; the maximal thermal neutron flux--5×10^{14} n/cm^{2}sec.

The fuel assemblies of the reactor, consisting of concentrically arranged tubular fuel elements with separating lengthwise fins, are located in the channels with ordinary pressure water cooling.

Water is fed to the channels from top, through a system of collectors; each collector distributes it between several chan-

nels. Water removal is effected below, from each channel separately.

Within the limits of the core, the height of which is 1 m, the channels run through the beryllium blocks having the shape of a hexahedral prism. The interval between the channels is 150 mm.

Beryllium blocks are also used in the first layer of the side reflector; the second layer of the reflector consists of aluminium-coated graphiteb blocks.

Sixteen uniflow loop channels of a V-shaped design, up to 150 mm. in diameter, can be installed in the core and the reflector.

The core and the working and loop channels of the reactor, as well as the collectors and tubular branches are completely sumbmerged into a pool.

The beryllium and graphite blocks are cooled with water from the pool, which circulates through a separate circuit.

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Abstract

POTENTIALITIES OF PULSED REACTORS

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A pulsed reactor differs from the common type in that it operates in cycles, the cycle including the operation and cooling stages. Within certain limits, the higher the cooling time to operation time ratio (duty ratio) the cheaper is the realization of the desired neutrons flux. There are the following types of pulsed reactors: the pulsed type proper, the cycle pile, and the homogeneous reactor.

The difference between the two latter types is to a certain degree a conditional one - in the cycle pile the core moves along the reactor body, while in the homogeneous type the situation is reversed.

The pulsed reactor proper is advantageous in experiments including neutron sorting by the time of flight. In this case narrower but sufficiently frequent pulses are desirable. Such a reactor with a cooling system removing several megawatts of power and with pulse frequency of the order of 10^2 per second, have advantages over the stationary high neutron flux reactor since it uses tens of times less fuel than the latter and its heat transfer system is relatively cheap.

The cycle pile is preferable to the pulsed reactor when a strong integral flux is required along with high flux. Such is the case, for instance , in experiments on neutrino interaction with matter where the neutrino source is a neutron-activated isotope. A great integral flux

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is necessary to obtain good statistics while a high duty ratio (with a given mean power) provides a favourable relation between the background and the effect.

In the magnitude of neutron fluxes obtainable, pulsed reactors proper are not superior to cycle piles or homogeneous reactors. In all cases the ultimate neutron density is approximately equal to energy, which can be imparted to a unit volume of a coolant during one cycle of heating (εN), divided by fission energy, E_f. To be more exact,

$$n_{max} = (\nu_f - 1) \frac{\varepsilon}{E_f} N ,$$

where N is the number of atoms in a unit volume of the reactor. If there is no provision for active zone transition into a plasma state ($\varepsilon \gtrsim 0.3$) , then $n_{max} \gtrsim 10^{14}$ cm^{-3}.

Pulsed reactors with different fuel concentrations have (for a given εN) different pulse durations and integral flux densities, however these quantities' ratio has the above limit.

This limit also holds for homogeneous reactors with one-way movement of the coolant, though at first glance limitless forcing is possible by accelerating the coolant speed. Actually, at speeds of the order of 10^2 m/sec the carrying away of neutrons by the coolant becomes serious. To maintain the reactor at critical it is necessary to increase the size of the core. To a certain limit the required height of the core increases proportionally to the speed of the coolant, the heat release in a unit volume of the reactor tending to its final limit. The limit can be shifted, in principle, only by the use of a counter current of coolant streams. Thus to shift the limit by an order, streams of a cross section not exceeding a quarter of the reactor's cross section are required.

The paper discusses prospects for the construction of pulsed reactors which would open up new horizons for

study of neutrino interactions with nucleons and electrons.

Several possible versions of such reactors are discussed and compared, and neutron physics and engineering aspects of the problem studied.

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Abstract

CONSTRUCTION OF RESEARCH REACTOR MP FOR TESTING FUEL ELEMENT AND MATERIALS

Goncharov V.V., Babulevich E.N., Nikolaev Yu.G. et al

The report describes a new research reactor MP of an original design. This reactor was built in the Atomic Energy Institute named after I.V. Kurchatov for testing fuel elements and materials. It combines the advantages of a pool-type reactor and those of a channel-type reactor.

The reactor power is equal to 20 MW. The maximum thermal neutron fluxes on uranium and in the neutron trap equal $2 \cdot 10^{14}$ n/cm^{2}sec and $6 \cdot 10^{14}$ n/cm^2 sec respectively.

Highly enriched uranium is used as a fuel.

The reactor fuel assemblies consist of several thin-walled, three-layer, tubular fuel elements each provided with three longitudinal spacing ribs; the central or fuel layer of a fuel element has a uniform thickness.

The fuel assemblies are loaded into the channels and are water-cooled under pressure.

The channels form a square lattice of a spacing widening in the bottom-to-top direction, i.e. all the reactor channels, except the central one, are inclined.

Within the reactor core of 1 m heigh, blocks made of metallic berillium are installed between the channels; the side reflector is assembled of graphite blocks enclosed in an aluminium can. The heat-transferagent (ordinary water) is conveyed to and drained from each channel from the top.

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The core, channels, water coolant tubes and their collectors are submerged in the water pool.

Besides the first circuit with water under pressure for cooling fuel elements, there is another circuit in which the pool water circulates to cool the berillium and graphite blocks of the core and reflector stacking.

Uniflow U - shaped loop channels are installed in nine points of the core and in four points of the reflector. The maximum diameter of the loop channel equals 150 mm in the core, and 220 mm in the reflector.

The reactor is also furnished with a number of experimental channels for irradiation of materials. Besides that, the material samples can be irradiated in the central part of the fuel assembly of any channel of the reactor.

The report gives the description of all main design units and technological systems of the reactor and its loop units.

The part of the report devoted to the description of the loop units sets forth the principal characteristics of the experimental loops of the MP reactor: water loop with a volumetric water boiling in the core; non-boiling water loop; water loop with volumetric water boiling in the core and with the equipment made of carbon steel; loop with organic coolants, gas loop with carbon dioxide coolant, gas loop with helium coolant.

The report provides the reactor physical and technical characteristcs obtained during its physical start-up and in the course of reaching its rated capacity.

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Abstract

THE EXPERIENCE OF THE PULSED FAST REACTOR OPERATION AND ITS CHARACTERISTICS AT INJECTION OF NEUTRONS FROM A MICROTRON

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Shapiro F.L.

In this report the experience of the three-year operation of the pulsed fast reactor is described. The main disadvantage of the PFR, as a pulsed source for neutron spectrometry, is a long duration of a burst. In this connection recently the work has been performed on the construction of the injector of fast neutrons multiplying in the PFR operating in subcritical mode. The use of the subcritical system for multiplying neutrons from a pulsed accelerator comes across some difficulties connected with delayed neutrons multiplication. In the pulsed reactor the multiplication coefficient reaches some large magnitude only for a very short period of time. As a result, multiplication takes place only for prompt neutrons, while the average magnitude of the multiplication coefficient for delayed ones is small. This permits to use the PFR for the neutron multiplication. A burst is initiated by photoneutrons from bremsstrahlung of fast electrons generated by a microtron. In this report some peculiarities of such a system in different conditions of operation are considered.

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Abstract

THE WORK AND THE EMPLOYMENT OF THE RESEARCH REACTOR WWR-M WITH THE MAXIMUM NEUTRON FLUX 3.10^{14} $\frac{n}{cm^2\ sec}$

D.M. Kaminker, K.A. Konoplev et al.

Reactor WWR-M is a modification of the **serial** water-water reactor WWR-C which had the neutron flux 10^{13} $\frac{n}{cm^2\ sec}$ The maximum neutron flux 3.10^{14} $\frac{n}{cm^2\ sec}$ can be obtained in the water trap which is placed in the active zone of the reactor. While employing the reactor WWR-M there accumulated a considerable experience in its exploitation with a high neutron flux.

The increase of the power of the reactor demanded the creation of such additional systems as a closed aired set, a semiautomatic dosimetrical control and others.

Horizontal beams of the reactor WWR-M are applied for the research in nuclear physics and in solid state physics. Specific organization is necessary for each experiment. Some common conditions for the research with high neutron beams are worked out.

After power work of the reactor five additional canals were hollowed in biological shield for experiments in **tangential** geometry of beams.

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Abstract

SPECIFIC FEATURES OF HEAT EXCHANGE IN NON COMPRESSIBLE NON-NOILING LIQUID-COOLED REACTORS

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The paper gives the results of a theoretical study of specific features of heat exchange processes in the channels of non compressible liquid-cooled reactors.

The author discusses the possibilities of extending Prandtl's hypothesis about turbulent transfer coefficients to arbitrary flows. Approximate equations for rate and temperature pulsations in an arbitrary liquid flow are derived from formulae written as integral equations describing the transfer of rate and temperature pulsation components. A mathematical analysis of "mixing lengths" introduced by Prandtl is presented.

The author considers the problem of stabilized heat transfer and hydraulic resistance in the reactor channels. The approximation of the turbulent viscosity coefficient is proposed used to calculate stabilized heat transfer and hydraulic resistance in tubes and for streamline flows parallel to rod bundles. The results of calculations in a broad range

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of Reynolds and Prandtl numbers are in good agreement with the available experimental data. The use of the hydraulic diameter for heat transfer and hydraulic resistance calculations is shown to be invalid for the case of streamline flows parallel to bundles of rods. A new method is suggested for determining the equivalent diameter of rod bundles which makes it possible toa apply the tube flow formulae for calculating the heat transfer and hydraulic resistance coefficients in rod bundles.

The paper presents calculations of heat transfer in circular tubes in the initial thermal region and in the sinesection thermal loop- Relaxation length values for the temperature curve are given which help determine heat transfer for an arbitrary thermal loop.

The author discusses the problems of critical thermal loads during surface boiling of heat-transfer agents in the reactor channels and the mechanism responsible for a heat-exchange critical point. An analitical expression is developed for critical thermal loads for tube flows below boiling point. The equation derived well agrees with the experimental data.